Exelon Generation.

June 20, 2014

10 CFR 50.73

SVP-14-048

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Quad Cities Nuclear Power Station, Unit 2
Renewed Facility Operating License No. DPR-30
NRC Docket No. 50-265

Subject: Licensee Event Report 265/2014-002-01, "Cable Tray Fire Caused by Non-Conforming Cable Routing"

Enclosed is Licensee Event Report (LER) 265/2014-002-01, "Cable Tray Fire Caused by Non-Conforming Cable Routing," for Quad Cities Nuclear Power Station, Unit 2. This revised LER provides that an engineering analysis was performed that demonstrated this event did not constitute a safety system functional failure.

This report is submitted in accordance with 10 CFR 50.73 (a)(2)(iv)(A) which requires the reporting of any event or condition that resulted in manual or automatic actuation of any of the systems listed in paragraph (a)(2)(iv)(B); in accordance with 10 CFR 50.73 (a)(2)(v)(D) which requires the reporting of any event or condition that could have prevented the fulfillment of the safety function of structures or systems that are needed to mitigate the consequences of an accident; and in accordance with 10 CFR 50.73(a)(2)(i)(A), which requires the reporting of the completion of any nuclear plant shutdown required by the plant's Technical Specifications.

There are no regulatory commitments contained in this letter.

Should you have any questions concerning this report, please contact Mr. W. J. Beck at (309) 227-2800.

Respectfully,

Ken OL (for)

Scott Darin
Site Vice President
Quad Cities Nuclear Power Station

cc: Regional Administrator – NRC Region III
NRC Senior Resident Inspector – Quad Cities Nuclear Power Station

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NRR

NRC FORM 366 (02-2014) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB: NO. 3150-0104 EXPIRES: 01/31/2017
LICENSEE EVENT REPORT (LER) (See Page 2 for required number of digits/characters for each block)	Estimated burden per response to comply with this mandatory collection request: 80 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the FOIA, Privacy and Information Collections Branch (T-5 F53), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to Infocollects.Resource@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

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4. TITLE
Cable Tray Fire Caused by Non-Conforming Cable Routing

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MONTH	DAY	YEAR	FACILITY NAME N/A	DOCKET NUMBER N/A
04	02	2014	2014 -	002 -	01	06	02	2014	FACILITY NAME N/A	DOCKET NUMBER N/A

9. OPERATING MODE	11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply)			
2	☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(i)(C)	☐ 50.73(a)(2)(vii)
	☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
	☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
	☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
10. POWER LEVEL	☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☒ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
	☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
008	☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
	☐ 20.2203(a)(2)(v)	☒ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(C)	☐ OTHER
	☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☒ 50.73(a)(2)(v)(D)	Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

LICENSEE CONTACT	TELEPHONE NUMBER (Include Area Code)
Tom Petersen – Regulatory Assurance	(309) 227-2825

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
B	SB	EXJ	N/A	Y					

14. SUPPLEMENTAL REPORT EXPECTED		15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☐ YES (If yes, complete 15. EXPECTED SUBMISSION DATE)	☒ NO		N/A	N/A	N/A

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)

On April 2, 2014, at 1228 hours, a Fire Alarm System (FAS) alarm was received for the Unit 2 D heater bay area. Although entry into the room at the time identified only a steam leak, subsequently various spurious alarms and electrical system anomalies occurred.

At 1303 hours, Unit 2 was manually scrammed, the turbine was tripped, and the main steam isolation valves (MSIVs) were closed to ensure the steam leak was isolated. A fire was identified to have occurred in the D heater bay (an area of the plant containing the high pressure (final stage) D feedwater heaters, and several Unit 2 cable trays and risers). The fire was extinguished by the automatic wet pipe sprinkler fire suppression system.

At 1340 hours, due to the manual de-energizing of safety-related motor control center (MCC) 29-1 in the reactor building in response to notification that smoke had been observed, an ALERT level Emergency Action Level classification was declared as HA3 (fire in a vital area affecting safety system equipment). The emergency was terminated at 2132 hours.

The cause of the event was an existing cable flaw that was caused by cable routing that exceeded the required minimum static bend radius.

Corrective actions included repairing impacted cables, replacing the failed steam seal expansion joint, operating procedure revisions, and additional inspections/tests.

The safety significance of this event was minimal. Given the impact on multiple systems, this report is submitted in accordance with 10 CFR 50.73 (a)(2)(iv)(A) for manual or automatic actuation of any of the systems listed in paragraph (a)(2)(iv)(B); in accordance with 10 CFR 50.73 (a)(2)(v)(D) for an event that could have prevented the fulfillment of the safety function of systems needed to mitigate the consequences of an accident; and in accordance with 10 CFR 50.73(a)(2)(i)(A), for the completion of a nuclear plant shutdown required by the plant's Technical Specifications.

NARRATIVE

PLANT AND SYSTEM IDENTIFICATION

General Electric - Boiling Water Reactor, 2957 Megawatts Thermal Rated Core Power

Energy Industry Identification System (EIIS) codes are identified in the text as [XX].

EVENT IDENTIFICATION

Cable Tray Fire Caused by Non-Conforming Cable Routing

A. CONDITION PRIOR TO EVENT

Unit: 2	Event Date: April 2, 2014	Event Time: 1228 hours
Reactor Mode: 2	Mode Name: Startup/Hot Standby	Power Level: 8%

B. DESCRIPTION OF EVENT

On April 2, 2014, Unit 1 was at full power and Unit 2 was in the process of starting up from a forced maintenance outage (approximately 8% reactor power). At 1228 hours with the reactor pressure at 920 psig and approximately two main turbine [TA] bypass valves [V] open, the main control room (MCR) [NA] received a Fire Alarm System (FAS) [IC] alarm [FRA] that smoke had been detected in the Unit 2 D heater bay (an area of the plant containing the high pressure (final stage) D feedwater heaters [SJ], and several Unit 2 cable trays [TY] and risers). The Station Fire Brigade Leader was dispatched to the D heater bay to assess the situation. Based on observations at that time, the Fire Brigade Leader reported to the MCR that there was no fire or smoke, but that there was a steam leak. This notification occurred approximately 10 minutes after the fire alarm had been received in the MCR.

At 1258 hours, per Operating procedure, Normal Unit 2 Startup, which requires the transition to the Run Mode between 4% and 12% reactor power based on Average Power Range Monitor (APRM) readings, the Unit 2 mode switch was taken to Run, placing Unit 2 in Mode 1. The Operating crew anticipated that subsequently allowing steam to the main turbine and rolling the turbine would assist in mitigating the previously observed steam leak. The MCR was unaware at this time of the exact source of the steam leak. There was no power ascension associated with this mode change at this time.

At 1302 hours, the MCR received numerous unexpected Unit 2 alarms and observed other anomalous indications on the MCR panels. Based on these alarms and indications, and since there was a known steam leak in Unit 2, the Unit was manually scrammed at 1303 hours. All control rods fully inserted as expected. A turbine trip was initiated at 1305 and the main steam isolation valves (MSIVs) were closed at 1312, isolating the previously observed steam leak.

At 1307 hours, in response to field reports that sparks had been observed in the turbine building ground-floor hallway outside the Unit 2 D heater bay, MCR Operators actuated the plant fire siren and dispatched the Fire Brigade and an additional Fire Brigade Leader to assist with required Fire Brigade Leader duties.

At 1315 hours the MCR received a report that smoke was now observed at the Unit 2 hydrogen seal oil vacuum pump breaker cubicle in MCC 28-2 (motor control center) [MCC].

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At 1317 hours the MCR received a report that thick black smoke had been observed in the D Heater Bay.

At 1321 hours the east side of the turbine FAS alarmed (a flow alarm [FA] in the fire system). This alarm was caused by the actuation of the fire suppression system [KP] (a fusible link sprinkler [SRNK]) in the D Heater Bay. The fire was extinguished by the automatic fire suppression in the area (i.e., sprinkler system).

At 1327 hours the MCR ordered MCC 29-1 (safety-related MCC in the Reactor Building [NG]) de-energized due to reports of smoke at that MCC.

At 1340 hours, due to the manual de-energization of MCC 29-1, an ALERT level emergency classification was declared by the MCR per Emergency Action Level (EAL) HA3 [fire in a vital area affecting safety system equipment].

At 1432 hours, ENS #49988 was made to the NRC under 10 CFR 50.72(a)(1)(i) due to the activation of an emergency classification; an ALERT was declared due to a fire in the Unit 2 turbine building. The notification was also made in accordance with 10 CFR 50.72(b)(2)(iv)(B) due to an RPS actuation while the reactor was critical, and in accordance with 10 CFR 50.72(b)(3)(iv)(A) due to valid system actuations.

After the Unit was stabilized and the fire and all smoke reports were mitigated, at 2132 hours the station emergency was terminated.

Several invalid actuations occurred during the event as a result of the fire, ensuing electrical transient, and de-energizing of equipment. These actuations included secondary containment, Group II (primary containment) and Group III (reactor water cleanup) isolations. In addition, the reactor recirculation pumps tripped on an invalid low reactor water level input.

As a result of the fire, various safety related equipment was de-energized and rendered inoperable.

Quad Cities Station is designed with control power transformers (CPTs) in various control circuits used in MCCs that are not fused and are ungrounded. Therefore, when a fault occurs on a cable, the potential exists for the CPT to become overheated. During this event, and as predicted by station calculations, various CPTs overheated and failed, producing smoke and heat damage inside their cubicles (examples include cubicles in MCC 28-2 and subsequently MCC 29-1 which resulted in the ALERT declaration). There were no cases observed in which flames were identified coincident with a failed CPT.

Given the impact on multiple systems, this report is submitted in accordance with 10 CFR 50.73 (a)(2)(iv)(A) which requires the reporting of any event or condition that resulted in manual or automatic actuation of any of the systems listed in paragraph (a)(2)(iv)(B) [manual RCIC actuation and RPS actuation/manual scram]; in accordance with 10 CFR 50.73 (a)(2)(v)(D) which requires the reporting of any event or condition that could have prevented the fulfillment of the safety function of structures or systems that are needed to mitigate the consequences of an accident [High Pressure Coolant Injection (HPCI) was inoperable]; and in accordance with 10 CFR 50.73(a)(2)(i)(A), which requires the reporting of the completion of any nuclear plant shutdown required by the plant's Technical Specifications (TS) [TS 3.0.3 was entered].

C. CAUSE OF EVENT

A root cause investigation Team was assembled to determine the cause of the fire and necessary corrective actions. The fire in the D Heater Bay was initiated when the high humidity and condensate from the steam leak provided the environment necessary for an existing flaw in an electrical cable [CBL] to fault to ground. Although the fire consumed the area of cable that contained the flaw, no reasonable mechanism for the fire was identified that did not include an initiating flaw in a cable. The root cause for this event (i.e., source of the cable flaw) was identified as routing of the Instrument Bus and Essential Service (ESS) Bus cables in a manner inconsistent with the current Exelon standard for

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minimum static bend radius for this type and size of cable. Specifically, the minimum static bend radius of 3.5 inches was not met. The routing for these cables (i.e., on the bottom of the tray, resting across rungs, and exiting at a sharp angle downward into a 40-foot vertical run) created a condition such that the mechanical stress on the cables could result in a failure before the end of its design life. This condition was introduced during original plant construction.

Although the cable was approximately 42 years old, no issues with the cable aging management program were identified. Visual inspection of like cables in the D Heater Bay, as well as testing at Exelon PowerLabs, indicated that the faulted cable insulation and jacket were typical of a 42 year old cable of this type, service and environment, with insulation properties (i.e., hardness and tensile strength) similar to cables in a milder environment (e.g., cable tunnel). The steam leak was determined to be due to a relief valve [RV] lifting in the main turbine steam [SB] seal (steam seal) system, and the failure of the downstream expansion joint [EXJ], which had been in a reversed orientation since original installation. The relief valve lifted due to incorrect operation of the steam seal system which was identified as a contributing cause. The incorrect installation (reversed orientation) of the expansion joint was also identified as a contributing cause.

D. SAFETY ANALYSIS

System Design

The purpose of the turbine gland sealing system is to prevent air leakage into, or radioactive steam leakage out of, the turbine shaft packings and turbine admission valve stem packings. The turbine gland sealing system is not credited for safe shutdown of the reactor or required to perform any reactor safety function. The turbine gland sealing system is designed in accordance with ASA B31.1 Code. The system is designed to provide low pressure sealing steam to the turbine shaft glands. The relief valve will maintain the system at a safe pressure if the control valves fail. Manual pressure control is possible using bypass valves. At low load, sealing steam is supplied from main steam via an air-operated control valve, which reduces the pressure to between 2.5 and 4.5 psig.

Safety Impact

Although this event involved an electrical fire / arc and subsequent insulation fire, the fire was extinguished by installed, automatic suppression (sprinkler system) prior to extending beyond the immediate area. The fire and associated cable damage caused breaker trips and the overheating of isolated CPTs. These effects did not adversely impact reactor shutdown, depressurization, and cool-down activities. There were no spurious actuations of equipment, and no additional fires were initiated beyond the top and bottom cable trays in the immediate area. In addition, although not required to extinguish the fire, the Station Fire Brigade was mobilized within minutes of notification of a fire, and controlled access to the area, coordinating the actions of Radiation Protection, Security and Off-site Firefighters. There were no human performance issues encountered during the unit shutdown or subsequent reactor depressurization and cool-down. Initiation and performance of the Emergency Response Organization was in accordance with plant procedures and the Emergency Plan. There were no injuries during the event.

With respect to 10 CFR 50 Appendix R fire assumptions, this fire was fully contained within the affected fire area, and since the station is designed to withstand a complete loss of one fire area, this fire did not challenge the assumptions for station Appendix R scenarios.

Although HPCI was declared inoperable during the event, a subsequent engineering analysis that was performed demonstrated the inoperable HPCI system did not constitute a Safety System Functional Failure (SSFF). (Reference NEI 99-02, Revision 7, Regulatory Assessment Performance Indicator Guideline, Section 2.2, Mitigating Systems Cornerstone, Safety System Functional Failures, Clarifying Notes, Engineering analyses.) As such, this event will not be reported in the NRC Performance Indicator (PI) for safety system functional failures since an engineering analysis was performed which determined that HPCI was capable of performing its safety function during this event.

NARRATIVE

Risk Insights

When reviewing this event against the Probabilistic Risk Assessment (PRA) for the station, this event resulted in a Conditional Core Damage Probability (CCDP) of less than 1E-06. This is not a significant impact on quantitative plant risk.

In conclusion, the overall safety significance of this event was minimal.

E. CORRECTIVE ACTIONS

Immediate:

1. Unit 2 was scrammed. The fire was extinguished.
2. Post-transient walk-downs were performed.
3. Equipment damaged by the fire was quarantined.
4. Developed cable inspection, testing, and repair strategy for damaged cables.

Follow-up:

1. A Root Cause Investigation was completed to determine the cause of the fire and required corrective actions.
2. Repaired cables that were damaged by the fire, or otherwise did not meet visual / test criteria (all safety related cabling was repaired).
3. Replaced and re-routed Unit 2 ESS and Instrument Bus cables out of Unit 2 D Heater Bay.
4. Replaced the failed steam seal expansion joint.
5. Inspected and tested the MCC 28-2 breakers associated with the IB Normal feed, ESS Bus Reserve feed, and the Transformer to ESS and IB feed.
6. Implemented revisions to the main turbine steam seal operating procedures to prohibit the use of the steam seal PCV bypass valve (2-3099-S2) above 250 psig reactor pressure.
7. Performed walk-downs to identify cables on Unit 2 that are in areas that are inaccessible at power, and that meet the three criteria (single conductor, riser to horizontal run, load/power above 50 amps), and evaluated any that did not meet the static bend radius specification.
8. Will perform a walk-down to identify cables on Unit 1 and 2 that are in areas that are accessible at power, and that meet the three criteria (single conductor, riser to horizontal run, load/power above 50 amps), and disposition any that do not meet the static bend radius specification.
9. During the next Unit 1 refueling outage, will perform a walk-down to identify cables on Unit 1 that are in areas that are inaccessible at power, and that meet the three criteria (single conductor, riser to horizontal run, load/power above 50 amps), and disposition any that do not meet the static bend radius specification.

F. PREVIOUS OCCURRENCES

The station events database, LERs, and INPO Consolidated Event System ICES (EPIX) were reviewed for similar events at Quad Cities Nuclear Power Station. This event was a cable tray fire caused by an existing cable flaw that was caused by cable routing that exceeded the required minimum static bend radius. There were no other previous similar occurrences identified at Quad Cities Nuclear Power Station that were associated with this type of failure.

G. COMPONENT FAILURE DATA

Failed Equipment: Main Turbine Steam Seal System Expansion Joint
Component Manufacturer: Farris Engineering
Component Model Number: N/A
Component Part Number: N/A

This event has been reported to ICES as Failure Report No. 311048.